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U. S. Nuclear Regulatory Commission
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LER 354/04-010-00
HOPE CREEK GENERATING STATION
FACILITY OPERATING LICENSE NO. NPF-57
DOCKET NO. 50-354

This Licensee Event Report entitled "Manual Reactor Scram Due to Moisture Separator Dump Line Failure" is being submitted pursuant to the requirements of 10CFR50.73(a)(2)(iv)(A).

Sincerely,

James Hutton
Plant Manager – Hope Creek

Attachment

BJT

C Distribution
LER File 3.7

IE22

NRC FORM 366 (6-2004) | U.S. NUCLEAR REGULATORY COMMISSION | APPROVED BY OMB: NO. 3150-0104 | EXPIRES: 06/30/2007

LICENSEE EVENT REPORT (LER)

(See reverse for required number of digits/characters for each block)

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4. TITLE
Manual Reactor Scram Due to Moisture Separator Dump Line Failure

5. EVENT DATE			6. LER NUMBER			7. REPORT DATE			8. OTHER FACILITIES INVOLVED	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REV NO.	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER
10	10	2004	2004	- 010 -	00	12	09	2004	FACILITY NAME	DOCKET NUMBER

9. OPERATING MODE: 1

10. POWER LEVEL: 69

11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR§: *(Check all that apply)*

☐ 20.2201(b)	☐ 20.2203(a)(3)(i)	☐ 50.73(a)(2)(i)(C)	☐ 50.73(a)(2)(vii)
☐ 20.2201(d)	☐ 20.2203(a)(3)(ii)	☐ 50.73(a)(2)(ii)(A)	☐ 50.73(a)(2)(viii)(A)
☐ 20.2203(a)(1)	☐ 20.2203(a)(4)	☐ 50.73(a)(2)(ii)(B)	☐ 50.73(a)(2)(viii)(B)
☐ 20.2203(a)(2)(i)	☐ 50.36(c)(1)(i)(A)	☐ 50.73(a)(2)(iii)	☐ 50.73(a)(2)(ix)(A)
☐ 20.2203(a)(2)(ii)	☐ 50.36(c)(1)(ii)(A)	☒ 50.73(a)(2)(iv)(A)	☐ 50.73(a)(2)(x)
☐ 20.2203(a)(2)(iii)	☐ 50.36(c)(2)	☐ 50.73(a)(2)(v)(A)	☐ 73.71(a)(4)
☐ 20.2203(a)(2)(iv)	☐ 50.46(a)(3)(ii)	☐ 50.73(a)(2)(v)(B)	☐ 73.71(a)(5)
☐ 20.2203(a)(2)(v)	☐ 50.73(a)(2)(i)(A)	☐ 50.73(a)(2)(v)(C)	☐ OTHER
☐ 20.2203(a)(2)(vi)	☐ 50.73(a)(2)(i)(B)	☐ 50.73(a)(2)(v)(D)	Specify in Abstract below or in NRC Form 366A

12. LICENSEE CONTACT FOR THIS LER

FACILITY NAME	TELEPHONE NUMBER (Include Area Code)
Brian Thomas, Licensing Engineer	856-339-2022

13. COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT

CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX
X	BJ	P	G080	N					

14. SUPPLEMENTAL REPORT EXPECTED		15. EXPECTED SUBMISSION DATE	MONTH	DAY	YEAR
☐ YES *(If yes, complete 15. EXPECTED SUBMISSION DATE)*	☒ NO				

ABSTRACT *(Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines)*

On October 10, 2004, at 1814 hours, the reactor recirculation pumps were taken to minimum speed and the reactor was scramed due to a steam leak (pipe failure) in the turbine building. Immediately following the scram, a reactor water Level 3 (+12.5") scram signal was received as expected due to water level shrink in the reactor vessel. As a result of the pipe failure, condenser vacuum degraded. Operators began to depressurize the reactor to transition water level control to the condensate system. Prior to establishing condensate feed control, a Level 2 (-38") reactor water level actuation was reached on the A and B channels. High Pressure Coolant Injection (HPCI) automatically started and Channel A and B isolations and actuations occurred. The main steam isolation valves (MSIVs) were manually closed by the operators prior to automatic closure due to degrading condenser vacuum. Another Level 3 scram signal occurred during plant stabilization while HPCI and Reactor Core Isolation Cooling (RCIC) were being used to control reactor pressure and level. During the cooldown of the plant using the safety relief valves (SRV), an additional Level 3 scram signal occurred due to level shrink when the SRV was closed. Hope Creek achieved cold shutdown on October 12 at 0509 hours.

The cause of the manual reactor scram was the failure of a moisture separator dump line to the condenser. Corrective actions, in part, include repair of the failed piping, procedure changes, development of an Operational and Technical Decision Making process, and Licensed Operator training concerning this event.

The above events are being reported in accordance with 10CFR50.73(a)(2)(iv)(A).

PLANT AND SYSTEM IDENTIFICATION

General Electric – Boiling Water Reactor (BWR/4)

Reactor Protection System {JC}*
High Pressure Coolant Injection System {BJ}*
Reactor Core Isolation Cooling System {BN}*
Reactor Water Clean Up System {CE}*
Residual Heat Removal System {BO}*
Radiation Monitoring System {IL}*

*Energy Industry Identification System {EIIS} codes and component function identifier codes appear as {SS/CCC}

IDENTIFICATION OF OCCURRENCE

Event Date: October 10, 2004
Discovery Date: October 10, 2004

CONDITIONS PRIOR TO OCCURRENCE

The plant was in OPERATIONAL CONDITION 1 (POWER OPERATION) at 100% prior to the pipe failure. At the time of the manual reactor scram, reactor power had been reduced to approximately 69%. There was no equipment out of service that impacted this event.

DESCRIPTION OF OCCURRENCE

On October 10, 2004, at 1739 hours, a condenser off-gas trouble alarm was received. At 1741 hours, a turbine building exhaust (TBE) radiation monitoring system (RMS) {IL} alarm was received with Operators noting a rising trend on the monitor. Operators were dispatched to investigate. An Equipment Operator (EO) (non-licensed) reported back at 1750 hours that there was steam evident on the 137' elevation of the Turbine Building. The Radwaste Operator reported at 1751 hours that condenser off-gas flow was elevated and rising. At 1759 hours, a power reduction was commenced to 80% power due to reports that the steam leak in the Turbine Building was continuing to degrade. The 6A feedwater heater extraction steam lines were isolated as the potential source of the steam leak, however, the steam leak continued to degrade. At 1814, the reactor recirculation pumps were taken to minimum speed and the reactor mode switch was locked in the shutdown position to scram the reactor. Immediately following the scram, a reactor water Level 3 (+12.5") scram signal {JC} was received as expected due to water level shrink in the reactor vessel. Operators began to reduce reactor vessel pressure using the turbine bypass valves to allow for use of the secondary condensate pumps for reactor water level control. During the pressure reduction, the B Reactor Water Cleanup (RWCU) pump {CE/P} tripped. Due to the continued degradation of condenser vacuum, the reactor feedwater pumps tripped. Actions were taken at this point to transition reactor vessel water level and pressure control to the High Pressure Coolant Injection (HPCI) {BJ} and Reactor Core Isolation Cooling (RCIC) {BN} systems. At 1817 hours, the Control Room Supervisor (CRS) directed the reactor operator to close the turbine bypass valves. As the bypass valves were closed, reactor water level reached the Level 3 scram setpoint (+12.5 inches) and continued to trend downward. RCIC was manually initiated to restore water level. As the turbine bypass valves were going closed, level continued to trend downward until reactor water level reached the Level 2 setpoint (-38") on two (A and B) of the four reactor vessel level channels. As a result of the Level 2 setpoint being reached on two of the reactor vessel level channels, the following equipment actuated:

DESCRIPTION OF OCCURRENCE (cont'd)

- HPCI automatically started and injected.
- BG-HV-F001 valve closed isolating RWCU.
- BB-SV-4310 valve closed isolating the reactor recirculation sample line.
- HB-HV-F004 and HB-HV-F020 valves closed isolating the drywell sump discharge lines.
- A Channel 1E breaker load shedding resulted in the trip of Emergency Air Compressor 10K100, Reactor Building Supply Fan 1BVH300, Reactor Building Exhaust Fan 1CV301, and Radwaste Exhaust Fan 0AV305.
- B Channel 1E beaker load shedding resulted in the trip of Reactor Building Exhaust Fan 1BV301, and Radwaste Exhaust Fan 0BV305.
- Filtration Recirculation Ventilation System (FRVS) vent fan AV206 started
- FRVS recirculation fans A213, B213, E213 and F213 started
- SK-HV-5018 and SK-HV-4957 valves closed isolating Drywell Leak Detection RMS
- EE-HV-4679, EE-HV-4652, EE-HV-4680, and EE-HV-4681 valves closed isolating Torus Water Clean Up (TWCU) and tripped the TWCU pump 0P229.
- KL-HV-5115 valve closed isolating Primary Containment Instrument Gas (PCIG) to the reactor building to suppression chamber vacuum breakers.

All of the above actuations were expected based on the actuation of the A and B channels.

With RCIC injecting into the vessel and water level increasing, HPCI injection into the reactor vessel was terminated.

Condenser vacuum continued to degrade. Prior to reaching the low condenser vacuum setpoint for automatic isolation of the main steam isolation valves (MSIVs), the operating crew manually closed the MSIVs and main steam line drains. Approximately 30 seconds following manual MSIV closure, the main condenser low vacuum setpoint for MSIV isolation was reached.

Following MSIV closure, the HPCI system was placed in service in pressure control mode for control of reactor pressure. While placing the HPCI system in pressure control, operators were not able to initially open the HPCI full flow test line valve (F008) when the open pushbutton was pressed. The reactor operator (RO)(Licensed) verified that the HPCI initiation signal was reset and depressed the closed push button for the HPCI injection valves. Depressing the closed push button cleared the interlock for opening the full flow test valve and HPCI was placed in the pressure control mode. The RCIC automatic flow control setpoint was reduced due to a rising water level in the reactor vessel. A few minutes later, RCIC was removed from service due to flow controller oscillations. Approximately 10 minutes later, RCIC was restarted with the reduced flow controller setpoint.

With the ROs controlling pressure and water level with HPCI and RCIC, the CRS ordered the Level 3 scram signal to be reset. At 1845, the reactor protection system scram signal was reset. At 1846, reactor water level dropped below the reactor water Level 3 scram sepoint (third Scram signal). The RO raised RCIC flow injection to restore vessel water level above the Level 3 scram setpoint.

At 1850, the CRS directed reactor pressure to be controlled in the 500-650 psi pressure band. The feedwater startup level control valve was then set in automatic at approximately 25" for transition of level control to the secondary condensate pumps.

With the plant stable, the reactor protection scram signal was reset at approximately 1903 hours. At approximately 2000 hours, turnover was conducted on station with the night shift operating crew.

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DESCRIPTION OF OCCURRENCE (cont'd)

At 2048 hours, the night shift CRS ordered the plant cool down to be recommenced using HPCI, RCIC or safety relief valves (SRVs). At approximately 2102 hours, the HPCI barometric condenser vacuum pump tripped on thermal overload. RCIC was subsequently placed in pressure control mode. The HPCI vacuum pump thermal overloads were reset at 2111 hours and RCIC was removed from service at 2117 hours. At 2124 hours, the HPCI vacuum pump thermal overloads tripped for the second time. At 2127, RCIC was placed in service in pressure control. At approximately 2131 hours, the operating crew decided to remove HPCI from service to prevent the spread of radioactive steam in the HPCI pump room with the loss of the barometric condenser vacuum pump. Shortly following the removal of HPCI from service, reactor vessel level reached Level 8 (+54") causing RCIC to automatically shutdown. Pressure control was then transitioned to the SRVs. The first SRV was opened to control pressure and reduce water level. At approximately 2156 hours, with reactor water level lowering, the operating crew closed the SRV when water level was approximately +25". Due to reactor water level shrink upon closing the SRV, reactor water level dropped to approximately +9" generating a Level 3 reactor scram signal.

At approximately 2203 hours, the reactor protection system Level 3 scram was reset. The feedwater startup level control valve setpoint was raised from 25 to 35 inches with reactor water level being supplied by the secondary condensate pumps to avoid subsequent Level 3 scrams. As a result, the Level 8 (+54") RCIC shutdown signal remained present due to the lowering indicated water level as the reactor vessel is cooled (i.e., Level 8 is monitored on the wide range level indication and Level 3 is monitored on the narrow range level indication). The SRVs were used to control reactor pressure with no further Level 3 scram signals.

During the plant transient on October 10, 2004, the A and B residual heat removal (RHR) loops {BO} were placed in suppression pool cooling at 1831 hours. Technical Specification Action Statement (TSAS) 3.6.2.3.b was entered for having both suppression pool cooling loops inoperable in accordance with Operations procedure guidance. With both loops inoperable, TSAS 3.6.2.3.b requires cold shutdown (Operational Condition 4) to be achieved within 24 hours. At 0509 hours on October 12, 2004, Hope Creek entered Operational Condition 4. Subsequently, a 1-hour report was made to the NRC on October 12, 2004, for the after-the-fact determination of an Unusual Event for failure to meet the shutdown requirements of TSAS 3.6.2.3.b. Further review of the event identified on October 11, 2004, at 1307 hours, the B RHR loop was taken out of suppression pool cooling mode and placed in standby. "B" RHR was subsequently aligned to shutdown cooling. As a result, TSAS 3.6.2.3.b to achieve cold shutdown within 24 hours no longer applied and TSAS 3.6.2.3.a became applicable. TSAS 3.6.2.3.a allows 72 hours to restore the inoperable train of suppression pool cooling to operable status. In addition, a follow-up analysis concluded that RHR should not be considered inoperable while in shutdown cooling mode; therefore, at least one loop of RHR was operable within 24 hours.

The above events are being reported in accordance with 10CFR50.73(a)(2)(iv)(A), any event or condition that resulted in the manual or automatic actuation of (1) Reactor Protection System (RPS) including reactor scram (2) general containment isolation signals affecting containment isolation valves in more than one system or multiple main steam isolation valves and (4) ECCS for boiling water reactors.

CAUSE OF OCCURRENCE

The cause of the manual reactor scram was the failure of the moisture separator dump line to the condenser. The moisture separator dump line failed where the 8" line enters the condenser due to 25 days of vibration (high dynamic loading) caused by flow through the failed open 1ACLV-1039A moisture separator high level dump valve. The original calculation for nozzle design did not consider dynamic loads for this piping. The crack occurred at the toe of the fillet weld of an encapsulation that was used to repair a crack that had occurred in 1988. Metallurgical analysis performed on the failed pipe section identified the predominant fracture mode as fatigue. The 1039A valve failed open and remained open for 25 days due to a pipe hanger that had become disconnected from the threaded eye nut and came to rest on the instrument tubing tray that supplied instrument air to the 1039A valve. The hanger wore a hole in the instrument

CAUSE OF OCCURRENCE (cont'd)

tubing causing the 1039A valve to initially open and then subsequently fully open. The hanger unthreaded from the eye nut as a result of vibration.

Operating procedures for moisture separator level control were inadequate to prevent extended operation with no water level in the moisture separator, which allowed two phase flow through the high level dump line. In addition, inadequate decisions were made by engineering and management to continue operating the moisture separator with the drain valve failed open.

The cause of the RCIC system flow oscillations is that the system operating procedures allowed operation of RCIC at low flow with the flow controller in automatic. The RCIC turbine control system design states that reducing the system flow below 75% of its rated flow value promotes the likelihood of control system instability. If control system instability occurs, it is recommended that the flow controller be put in manual mode.

The cause of HPCI vacuum pump tripping on thermal overload was due to the use of improper material to lubricate the shaft packing, which caused excessive friction. In addition, the packing gland follower was corroded and binding on the shaft sleeve. The improper lubricant (thread sealant) for the HPCI vacuum pump was listed on the lubrication screen. The lubrication screen is part of the SAP work management database and specifies the lubrication material for components in the plant.

The HPCI full flow test valve failed to open on initial demand due to the improper adjustment of a limit switch for HPCI feedwater injection valve HV-8278. The limit switch for the HV-8278 valve that makes up the interlock with the F008 valve was open even though the control room valve position indicated closed and the HV-8278 valve was actually in the closed position. This prevented the F008 full flow test valve from opening.

The Level 2 automatic engineered safety feature initiations and isolation signals were the result of the trip of the reactor feedwater pumps and the transition to the condensate system for reactor feedwater level control.

The Level 3 scram during plant stabilization was the result of less than adequate coordination of reactor level control, pressure control and the resetting of the reactor scram signal by the CRS and ROs. A decreasing water level trend existed prior to the scram reset.

The Level 3 scram that occurred during plant cool down while using SRVs for pressure control is attributed to the level control strategy implemented in the emergency operating procedures. The level control strategy in the emergency operating procedures to control level between Level 3 and Level 8 did not effectively support the use of the SRVs due to level indication differences between the narrow range and wide range indicators (i.e., Level 8 is monitored on the wide range level indication and Level 3 is monitored on the narrow range level indication).

The Level 8 automatic overfill protection turbine shutdown signals during the plant cool down were the result of the reactor water level control strategy executed to ensure adequate margin above the Level 3 scram setpoint.

PREVIOUS OCCURRENCES

A review of LERs at Salem and Hope Creek generating stations for the previous two years did not identify a similar event reported as a result of a pipe rupture necessitating a manual scram and closure of the MSIVs.

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SAFETY CONSEQUENCES AND IMPLICATIONS

The break of the moisture separator high level dump line to the condenser led to a manual reactor scram with loss of condenser vacuum. Operators appropriately closed the main steam isolation and drain valves manually prior to automatic isolation due to the loss of condenser vacuum. Although Operators were challenged with the control of reactor water level and pressure with the normal heat sink being isolated, reactor water level remained approximately 10 feet above the top of active fuel throughout the event. The SRVs were used in a controlled manner to bring the plant to cold shutdown.

Operators removed the HPCI pump from service during this event to limit the spread of radioactive contamination in the HPCI pump room; however, the HPCI pump was still capable of providing injection to the reactor core.

The peak instantaneous release rate of radioactivity from the pipe break in the turbine building was well below Technical Specification limits. The release from the pipe break was estimated at approximately 1.7% of the Technical Specification limit.

Based on the above, there was no impact to the health and safety of the public.

A review of this event determined that a Safety System Functional Failure (SSFF) has not occurred as defined in Nuclear Energy Institute (NEI) 99-02. Hope Creek was brought to a controlled safe cold shutdown condition following the moisture separator dump line failure. The equipment challenges identified during this event did not impact the ability to mitigate the consequences of an accident.

CORRECTIVE ACTION

1. The failed condenser nozzle associated with the moisture separator high level dump line will be repaired with an improved design to minimize stress intensification factors prior to startup.
2. The moisture separator drain line hanger will be installed in accordance with the installation recommendations of the hanger manufacturer.
3. The instrument tubing to the LV-1039A valve has been repaired.
4. Operations procedures for the moisture separators will be revised to prohibit extended operations with the LV-1039 valve open with a loss of level in the moisture separator drain tank prior to startup.
5. A formal process for Operational and Technical Decision Making (OTDM) will be established to apply effective decision making principles to management and technical decisions in response to plant conditions prior to startup.
6. Procedure HC.OP-SO.BD-0001, "Reactor Core Isolation Cooling," has been revised to warn of system control instability at less than 75% rated flow and to provide direction that the flow controller be placed in manual.
7. The HPCI vacuum pump/motor will be replaced and the pump will be greased using the proper lubricant.
8. The packing on the RCIC vacuum pump will be repacked and the grease replaced.
9. The lubrication screen data for the HPCI and RCIC vacuum pumps has been revised to reflect the proper lubricant.
10. A verification of lubrication screens has been performed to ensure that thread sealant was not specified in other lubrication applications. No other instances of improper use of the thread sealant were identified.
11. The HV-8278 valve limit switches were adjusted to ensure solid mechanical contact when the HV-8278 valve is in the closed position.
12. To address current motor operated valve testing during the ongoing refueling outage, a checklist was developed to structure the test result reviews with an additional independent engineering review of the test results.
13. Appropriate maintenance procedures for motor operated valve testing will be revised to provide clear criteria for limit switch contact adjustment and an engineering review of the test results.
14. The transient dynamics of the October 10, 2004, scram will be validated on the Hope Creek simulator to correct simulator model discrepancies and enhance the response of the model.

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CORRECTIVE ACTION

15. Classroom training will be developed for the licensed operators concerning the control of reactor water level during the October 10, 2004, scram. This training will be provided to licensed operators as part of Hope Creek startup training prior to performing watch-standing duties during critical operations.
16. Simulator training will be developed for the licensed operators concerning the control of reactor water level during the October 10, 2004 scram. This training will be provided to the licensed operators as part of Hope Creek startup training prior to performing watch-standing duties during critical operations.
17. A reactor vessel pressure and level control strategy has been developed for hot standby and cool down with the main condenser unavailable. Classroom and simulator training will be developed to train the licensed operators regarding the pressure and level control strategy prior to performing watch-standing duties during critical operations.
18. The simulator will be changed to model RCIC flow oscillations and turbine bypass valve response.

The actions specified above are being tracked in accordance with PSEG Nuclear's Corrective Action Program.

COMMITMENTS

This LER contains no commitments.